ARH-600 Background

- ARH-600 Criticality Handbook contains novel "data" presentations for idealized systems
  - Simple geometries
  - Pu, U-235 and limited U-233
- Based on early codes and cross section data
- Not validated to current expectations
- NCSP Task
  - Provide an electronic representation for curves (CritView)
  - Add comparative data (MCNP, SCALE and other Handbook data)

ARH-600
Major FY2010 Activities

- ARH-600 Parameter Sensitivity Study
- MCNP model calculations matching ARH-600 curves
- Inclusion of MCNP Data in CritView Database

ARH-600 Parameter Sensitivity Study

- Precise values used in original modeling unknown
- Plutonium Density – Reasonable variation had minimal impact on results
  - 19.6 g/cc assumed; +/- 2% in density resulted in ~+/- 0.3% in MCM
- Definition of "Full" Water Reflection – Deemed unlikely to be a source of discrepancy
  - 10" used in MCNP calculations; little change in MCM found beyond 6" of water reflection for Pu + water systems

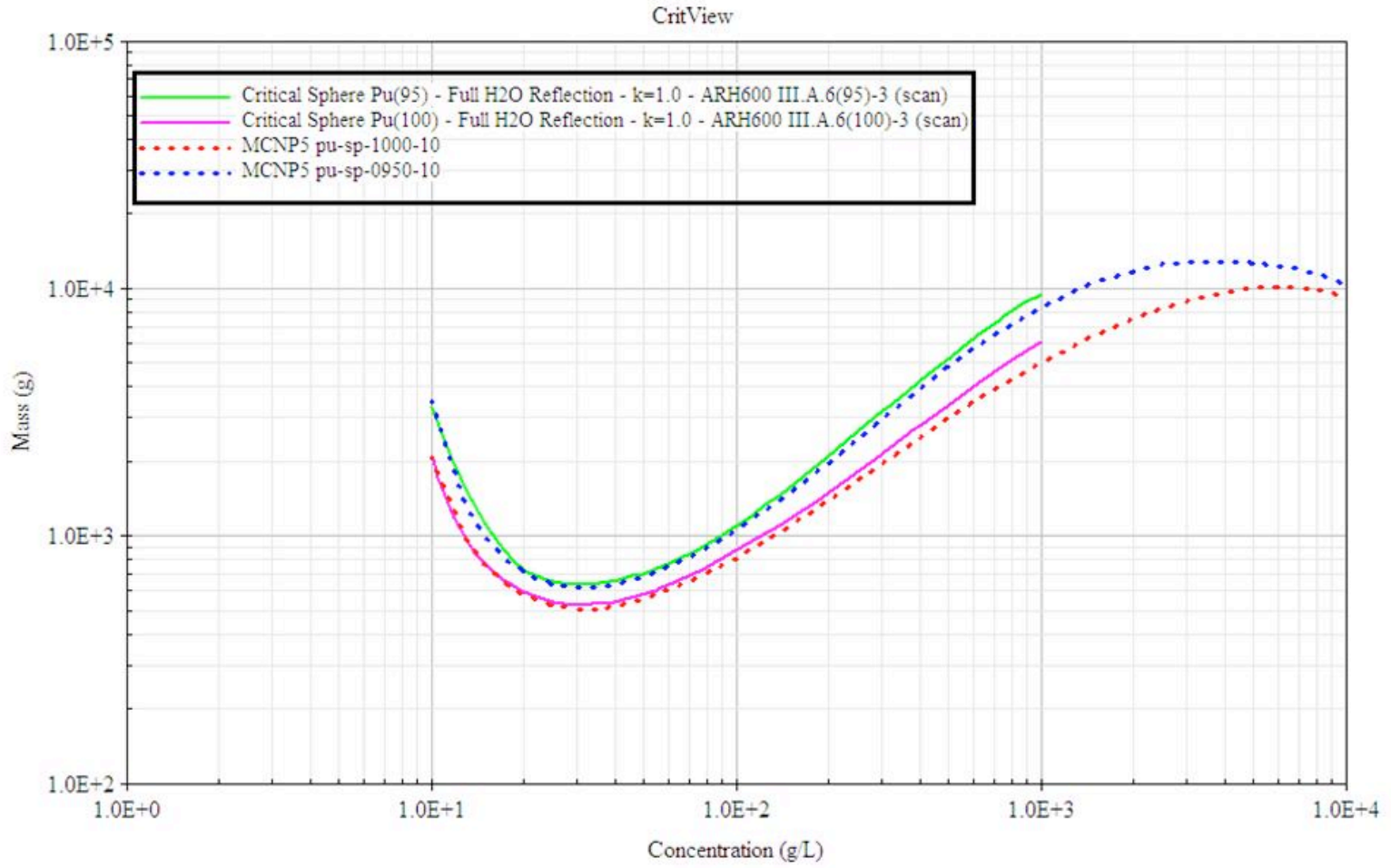
ARH-600 - MCNP Modeling to Reproduce ARH600 Data

- Over 60,000 individual MCNP cases run
  - Most major subsets of ARH600 included
  - Fissile material includes: Plutonium/Uranium/Uranium Oxides
  - Geometries include: Spheres, infinite cylinders, infinite slabs
  - Reflector cases include: Unreflected, 1” Water Reflected and Full (10”) Water Reflected
  - Did not include Nitrate Solutions
- Each case evaluated over a range of concentrations and diameters in order to produce k_{eff} curves that ranged from ~0.93 to ~1.01

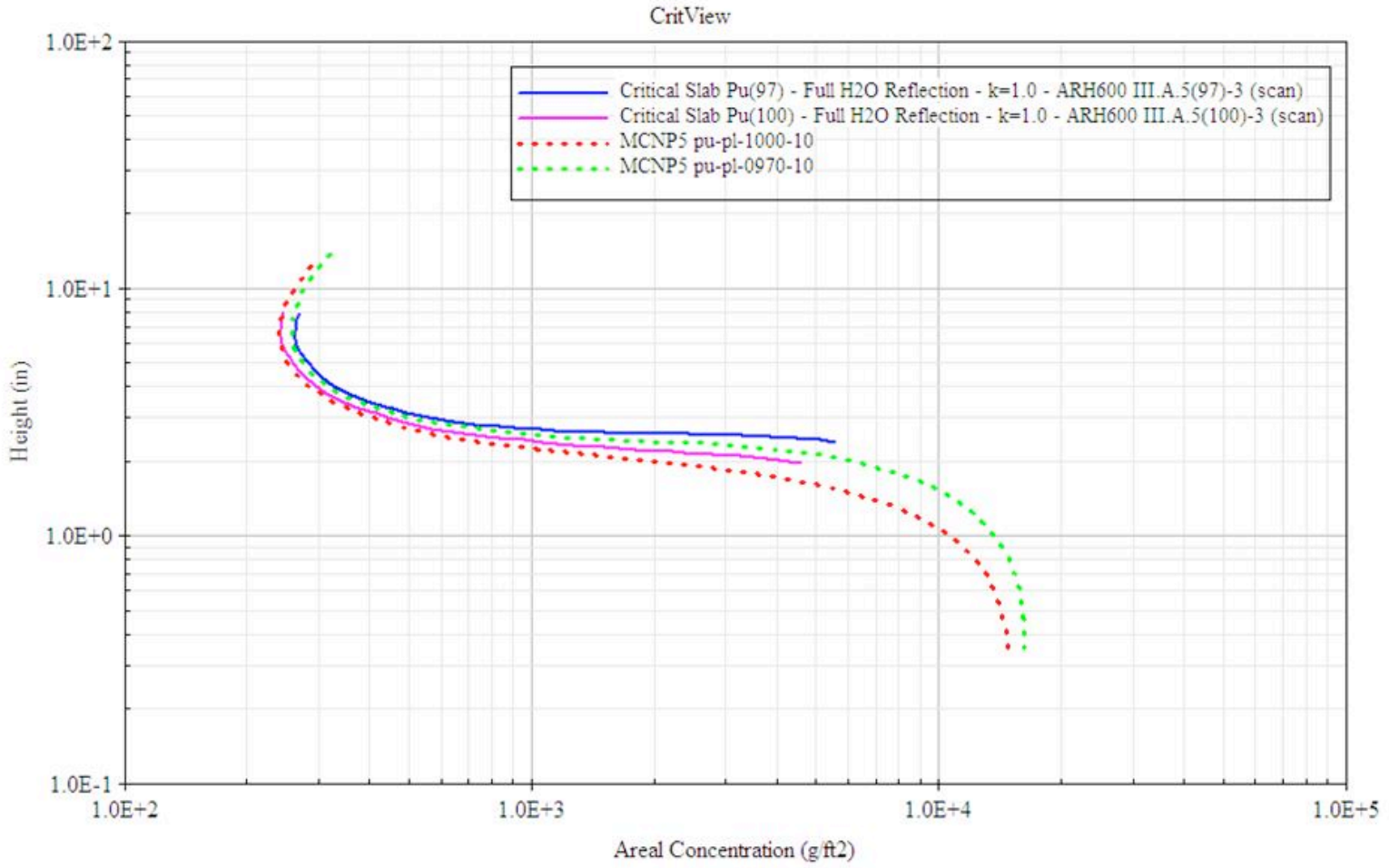
ARH-600 – Inclusion of MCNP Data into Critview Database

- Added k_{eff}=1.0 subset of results to CritView database
- Approximately doubled the size of the CritView database
- New database and documentation of calculations added to NCSP Web Site

ARH-600
Data Comparisons



ARH-600
Data Comparisons



ARH-600
FY2011 Planned Activities

- Issue peer reviewed report documenting added MCNP calculations for remaining ARH-600 curves
- Issue a peer reviewed report documenting added LA10860 Data for CritView libraries
- Provide updated CritView data libraries and documentation to LLNL for posting on the NCSP website

Actinide Integral Cross Section Progress - Background

- New Activity Proposed for FY2010
- Task Team (Hanford, SRS and ORNL)
- Task Focus
  - Extraction of relevant integral cross section data for actinides
  - Investigation of potential use of data to support burn-up code validation
  - Identification of irradiated targets available for investigation

Actinide Integral Cross Section Progress – FY2010

- Survey of Hanford and Savannah River published literature
  - Used publicly released reports and documentation
  - Pertinent Hanford and Savannah River Plant documentation containing neutron physics measurement results
  - Collected pertinent reports containing isotopic data at various exposures and reactor design features
  - Sorted 70 data sets into manageable configurations
  - SRS review did not identify additional relevant data

Actinide Integral Cross Section Progress – Example Report Listing

Reference Listings U-Pu Trees

IDENT U-Pu	Pub. Date	Doc. No.	Doc. Title	Authors	Elements Covered	Analysis/Data
1	9/12/1986	UNI-2592*	^{238}Pu Production from Initial ^{236}U in MKIV Fuel Based on Measurements	Wittekind, WD Clayhold, J	U, Np, Pu	Yes/Yes
2	6/10/1968	DUN-4349*	Effective ^{238}Pu and ^{237}Np Neutron Absorption Cross-Sections in N and K Reactors	Robyler, SP	U, Np, Pu	Yes/No
3	7/31/1967	DUN-2913*	Final Report PT-68 Results of N-Reactor Neptunium Irradiation	Constable, DW	Np, Pu	Yes/Yes
4	4/12/1967	RL-GEN-1599	Preliminary Results of the First Neptunium Target Column	Constable, DW Curtiss, DN	Np, Pu	Yes/Yes
5	12/15/1986	HEDL-810619*	Results of Analyses of PUREX D5 Tank Samples and Composites for Characterization of N Reactor MK-IV Spent Fuel Batch - Key 14832	Matsumoto, WY	U, Pu, Nd	No/Yes
6	2/25/1986	HEDL-810541*	Burn-up and Associated Analysis Results for N-Reactor Spent Fuel Sections - Data Sheets	Matsumoto, WY	U, Pu, Nd	No/Yes
7	12/2/1985	UNI-3771	N Reactor Isotopic Production Tables Calculated Adjustment of November 15, 1985 Using the Computer Code PTABLE2	Wittekind, WD	U, Np, Pu	Yes/Yes
8	10/23/1985	HEDL-810526	Results of Analysis of PUREX D-5 and A-3 Tank Samples to Characterize Key 14011 and Key 14306 N-Reactor Spent Fuel Batches	Matsumoto, WY	U, Pu, Nd	No/Yes
9	2/26/1985	UNI-3332	Production Table Predictions and PUREX Chemical Analysis	Wittekind, WD	Pu	No/Yes

Actinide Integral Cross Section Progress – Example Sorted Data

U-Pu Detailed Data																						
Reference Documents Numbers [1]			U-Pu	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20
Pre Irradiation Isotopes		U		)		)		)	)	)	)	)	)			)	)	)	)	)		)	)
a		U-234		)				)	)	)	)		)			)	)						
b		U-235		)				)	)	)	)		)			)	)	)	)	)		)	)
c		U-236		)				)	)	)	)		)		)	)	)	)	)	)		)	)
d		U-238		)				)	)	)	)		)			)		)	)	)		)	
e		Np-237				)	)	)			)		)		)	)			)	)	)		
		Cs or Ce					)	)	)														)
Post Irradiation Isotope		Np		)		)	)	)	)	)	)		)					)	)	)	)	)	)
		Pu		)		)	)	)	)	)	)	)	)					)		)	)	)	)
a		Pu-236				)						)	)							)	)		
b		Pu-238		)		)		)	)	)	)	)	)					)		)	)	)	)
c		Pu-239		)		)		)	)	)	)	)	)					)		)	)	)	)

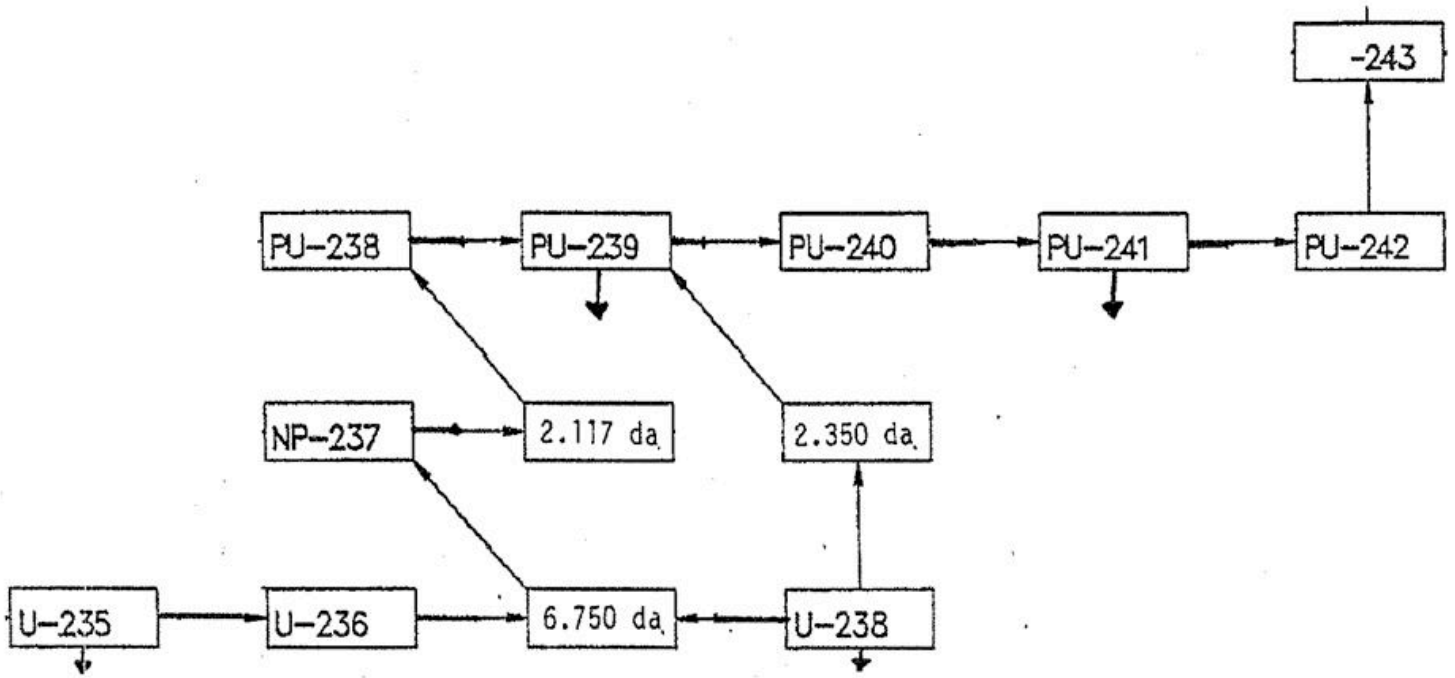
[1] The numbers from 1 - 20 refer to the documents in the reference listing for the U-Pu Tree

Actinide Integral Cross Section Progress – Typical Data Set

NUCLEAR FUEL BURNUP ANALYSIS RESULTS
[ND-148 IDMS METHOD]

LAB. NO.: / SAMPLE I.D.:		8045 UNC-8A0	8046 UNC-8BI	8047 UNC-8B0
U [GRAMS/G]	{1}	1.067E-01	3.846E-02	1.574E-01
U [GRAMS/G]	{1}	1.067E-01	3.846E-02	1.574E-01
U [ATOMS/G]	{1}	2.700E+20	9.729E+19	3.983E+20
U-234 WT %		0.009 { 1}	0.008 { 1}	0.009 { 1}
U-235 WT %		1.083 { 6}	0.855 { 6}	1.096 { 6}
U-236 WT %		0.062 { 1}	0.064 { 1}	0.060 { 1}
U-238 WT %		98.845 { 10}	99.072 { 10}	98.833 { 10}
PU [GRAMS/G]	{1}	1.297E-04	3.759E-05	1.820E-04
PU [ATOMS/G]	{1}	3.266E+17	9.468E+16	4.584E+17
PU-238 WT %		0.035 { 4}	0.038 { 4}	0.036 { 4}
PU-239 WT %		91.834 { 80}	93.640 { 80}	92.179 { 80}
PU-240 WT %		7.147 { 50}	5.633 { 50}	6.875 { 50}
PU-241 WT %		0.951 { 7}	0.671 { 7}	0.880 { 7}
PU-242 WT %		0.033 { 1}	0.018 { 1}	0.030 { 1}
FP-TOT [GRAMS/G]	{1}	1.854E-04	4.185E-05	2.551E-04
ND-148 [ATOMS/G]	{1}	7.993E+15	1.809E+15	1.100E+16
ND-143/148	{2}	3.23664	3.09571	3.25310
ND-144/148		2.60321	2.47299	2.60679
ND-145/148		2.19879	2.14060	2.20459
ND-146/148		1.74709	1.71613	1.74900
ND-150/148		0.42267	0.43825	0.42365
ND-148 E.F.Y.	{3}	1.686	1.691	1.686
PU-239 [FF]	{3}	0.09310	0.09783	0.08886
PU-240 [FF]		0.00000	0.00000	0.00000
PU-241 [FF]		0.00104	0.00076	0.00091
U-235 [FF]		0.85944	0.84177	0.86381
U-238 [FF]		0.04643	0.05964	0.04641
BURNUP CALCULATIONS:	{4}	UNC-N BASED	UNC-N BASED	UNC-N BASED
MWD/FISSION	{3}	3.765E-22	3.767E-22	3.764E-22
AVG. AT. WT. FISSIONED		235.563	235.620	235.545
BU ATOM % [ND-148]		1.750E-01	1.097E-01	1.633E-01
BU MWD/MTM [ND-148]		1.668E+03	1.045E+03	1.555E+03

Actinide Integral Cross Section Progress – U-Pu Tree



Actinide Integral Cross Section Progress – Proof-of-Principle

- Proof of principle test for extracting nuclear data for single set of data points
  - PTABLE2 code established to upgrade production tables
  - PTABLE2 solves boundary value differential equations for the U-Pu tree using 4th order Runge-Kutta method
  - Iterative technique used to adjust calculated isotopic cross sections to match analytic chemistry results
  - Effective one-group cross sections used to generate revised isotope production estimates
- Plan to modify PTABLE2 to provide integral cross sections for other isotopes

Actinide Integral Cross Section Progress – Irradiated Targets

- **Production Test PT-171 Irradiated Am-Al Targets**
  - **150 g Am (77.4 wt % Am-244 and 22.1wt% Am-243) combined with Al into 12 targets**
  - **Planned irradiation for 250 days in KE reactor**
  - **Estimated composition (12 targets): 17.3g Pu 239/Pu 241; 49.5g Pu 238; 15g Am 241; 32g Am 243; 4g Cm 244;**

    and 2g U (87% U 235)
  - **Targets currently stored in EBR-II cask on Hanford Site**

Actinide Integral Cross Section Progress – Future Work

- Select a particular problem and try to form a benchmark for LEU metal fuel
- Perform sensitivity analyses
- Demonstrate application for the Th-U and Am-Cm trees
- Attempt to use three energy group cross-sections to fit data
- Document the results and preserve the reference documents

Anomalies Report Background

- Last "Anomalies of Criticality" Rev. 5 published in 1979
- Duane Clayton and Staff at the Critical Laboratory continued to compile anomalies data
- NCSP funded activity to publish the new anomalies data developed between 1979 - 2000

Anomalies Report Development Overview

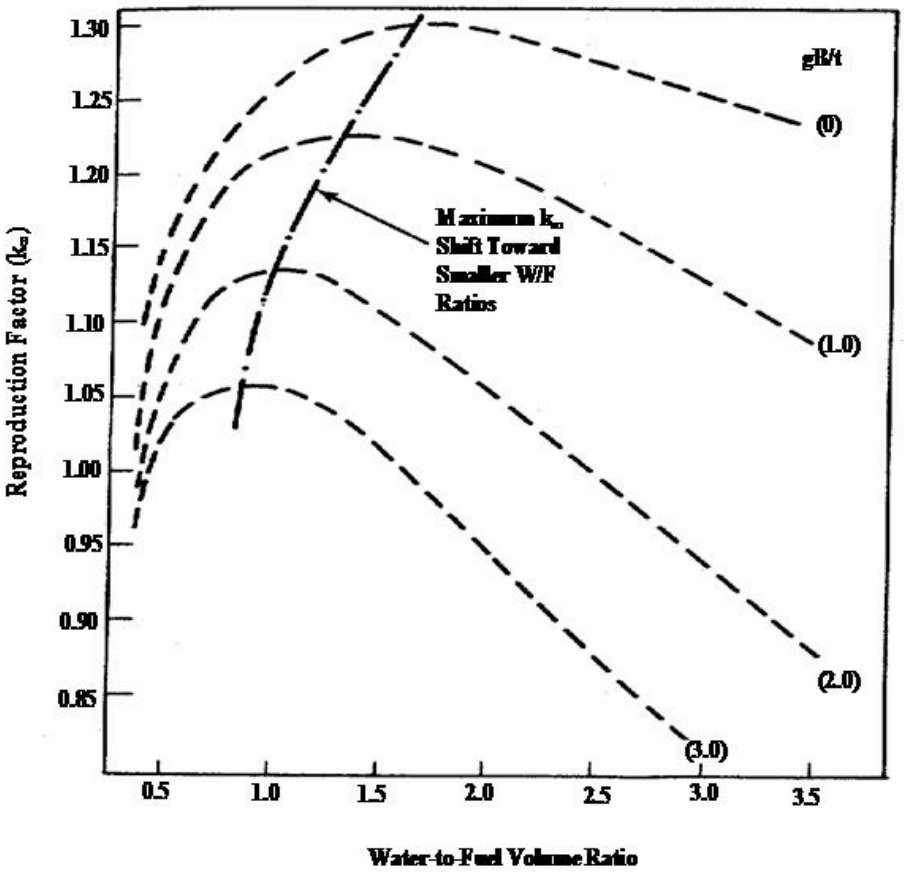
- Team established to collect new "anomalies" and issue revised document
  - Andy Pritchard and Mike Durst (PNNL), David Erickson and Ray Puigh (Hanford)
- Revision 6 reviewed by Duane prior to publication
- PNNL published in February 2010
- Document available on NCSP website

Anomalies Report
Major Changes in Revision 6

- Physical characteristics of the actinides
- Safety implications of anomalous effects of neutron absorbers on criticality
- Interstitial moderation and its importance to criticality
- Geometry effects
- Universally safe containers
- Super-heavy elements and an island of stability beyond Californium

Anomalies Report - Boron Use for Criticality Control of Fuel Lattices

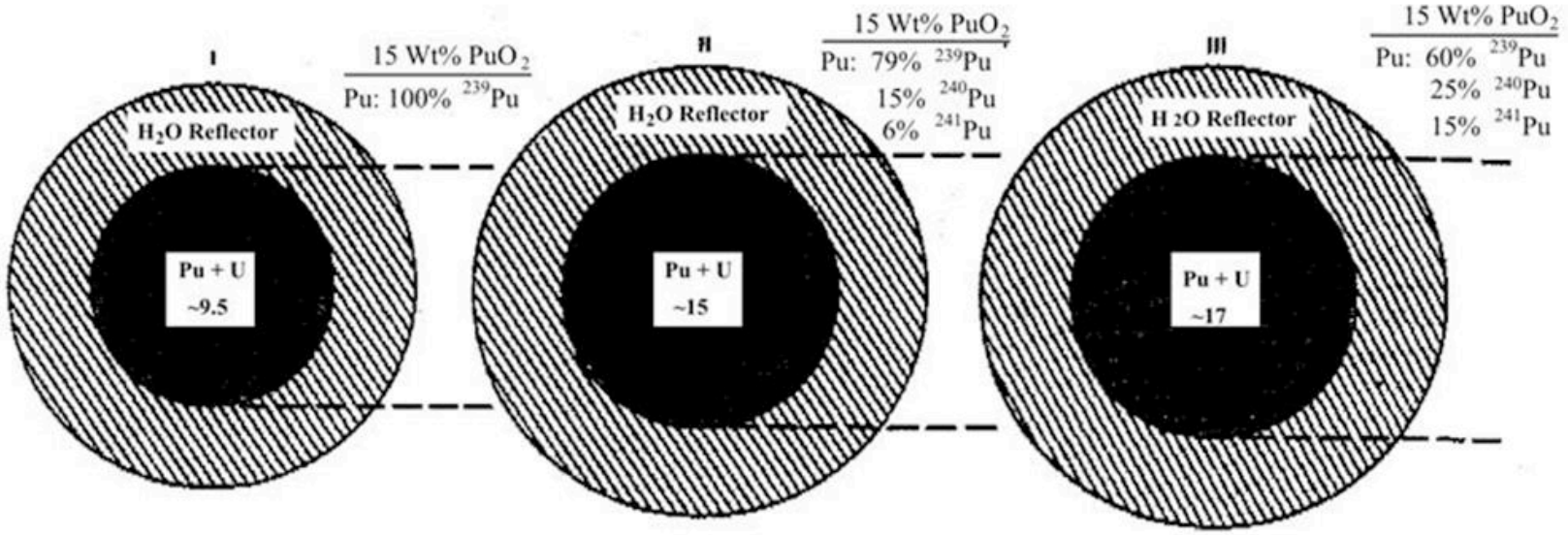
- High concentrations of thermal neutron absorbers can shift neutron energy
- Peak k_{inf} shifts to lower water-to-fuel ratio
- Compacting fuel rod lattice from optimum spacing in water could result in smaller critical size for heavily borated lattices



Anomalies Report – Effect of Pu Isotopes on Minimum Critical Size

Effect on Minimum Critical Size as Content of Heavier Isotopes of Plutonium is Increased

$$\frac{PuO_2}{PuO_2 + U(Nat.)O_2} = 0.15$$



For 15 Wt% PuO_2 in mixed oxide, minimum critical size continues to increase as higher isotopes of Pu displace ^{239}Pu; smallest size occurs for case 1 and largest for case III.

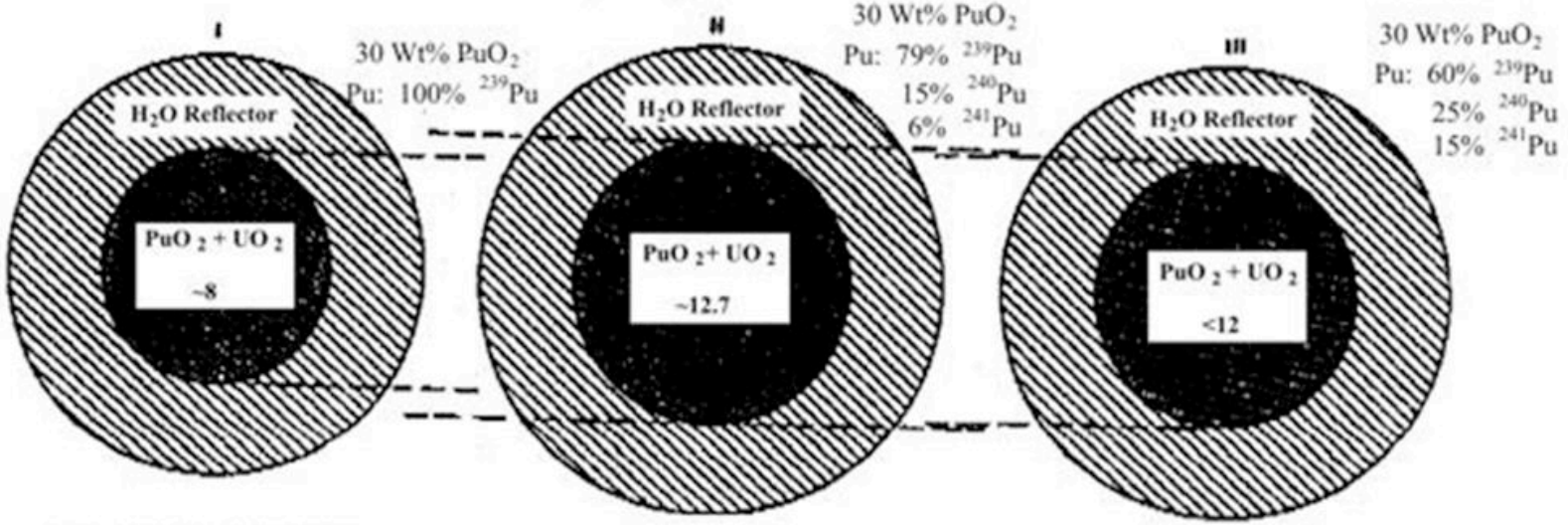
Anomalies Report– Effect of Pu Isotopes on Minimum Critical Size

- In the absence of ^{240}Pu, the minimum critical dimensions occur for the heterogeneous systems under well moderated conditions
- If the Pu content in the natural U is substantial, minimum critical dimension is obtained under low or essentially unmoderated conditions
- At high Pu content and a relatively fast neutron spectrum, the ^{240}Pu begins to fission in substantial quantity and contributes neutrons to the chain reaction

Anomalies Report– Effect of Pu Isotopes on Minimum Critical Size

Calculated Variation in Minimum Critical Volume for Heterogeneous Systems of Mixed Oxides as Function of ^{240}Pu Content in Pu

$$\left(\frac{PuO_2}{PuO_2 + U(Nat.)O_2} = 0.30 \right)$$



Minimum Volume at H/Fissile ≃ 70

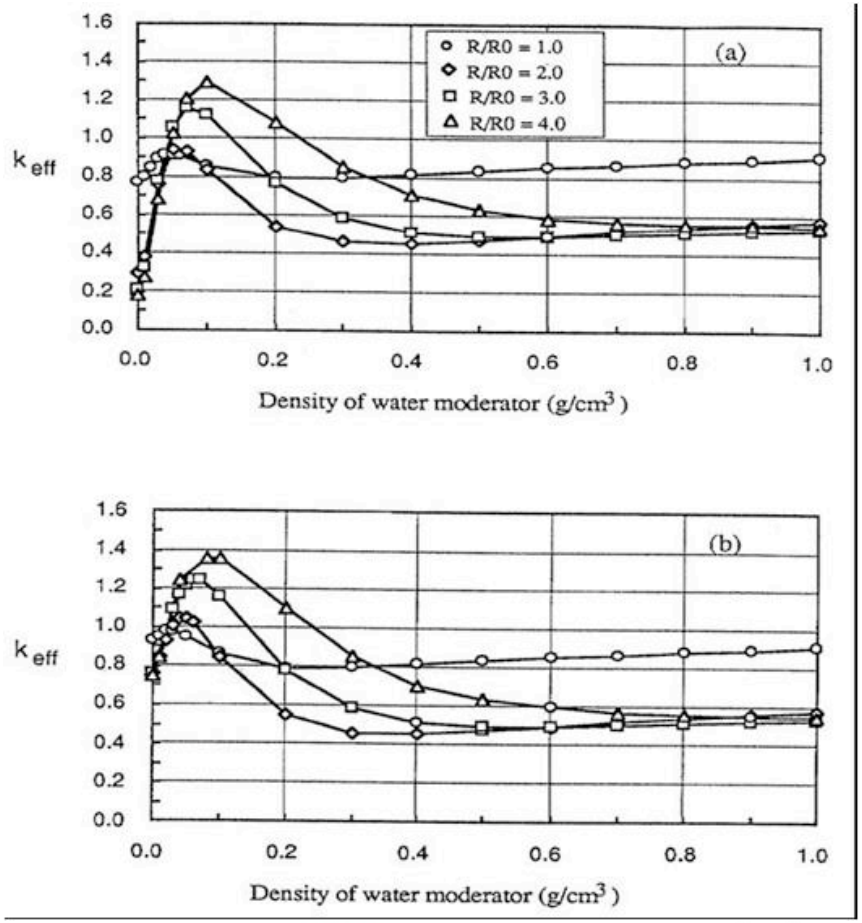
Minimum Volume at H/Fissile ≃ 100

Minimum Volume at H/Fissile < 8

Anomalies Report - Reflector Effects on Lattices

- **Unreflected lattices significantly subcritical with no interstitial water**
- **Little difference between unreflected and reflected lattices with some interstitial water**
- **Interstitial water reduces neutron leakage from system**

10^3 Arrays of Dry 15-kg 235-U Units
60.96-cm CTC Separations:
(a) unreflected; (b) Full water reflected



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